April 14, 1986

DCR 016

Docket Nos. 50-317
and 50-318

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

Dear Mr. Tiernan:

The Commission has issued the enclosed Amendment Nos. 117 and 99 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 applications dated February 22, 1985 and October 25, 1985.

The amendments change the Unit 1 and Unit 2 Technical Specifications (TS) to: (1) revise the Basis for the Containment Isolation Signal (CIS)/Safety Injection Actuation Signal (SIAS) setpoint for containment high pressure in TS Basis 2.2.1, "Reactor Trip Setpoints"; (2) change the allowable scheduling for moderator temperature coefficient (MTC) determination as required by TS 4.1.1.4.2c, "Moderator Temperature Coefficient"; (3) require that two charging pumps, required to be operable above 80% power, each be provided with an independent power supply per TS 3.1.2.4, "Charging Pumps - Operable" (Unit 1 only); (4) provide for additional channels associated with measurement of containment water level and change the statement regarding implementation of remedial actions in TS 3/4.3.3.6, "Post-Accident Instrumentation"; (5) correct a syntax error in TS 3.4.4, "Pressurizer" and a spelling error in TS 3/4.6.1.1., "Containment Integrity"; (6) update and clarify the reporting requirements of TS 6.9.2, "Special Reports"; (7) delete the Surveillance Requirements of TS 4.5.2g, "ECCS Subsystems T_{avg} [greater than or equal to] 300°F" - and redesignate the remaining Surveillance Requirements; (8) delete the reference to the 1971 Edition of the ASME Boiler and Pressure Vessel Code in TS Basis 3/4.7.1.1, "Safety Valves"; (9) delete a seismic sway arrester (snubber) from the operability and Surveillance Requirements of TS 3/4.7.8, "Snubbers" (Unit 1 only); (10) replace a reference in TS Basis 3/4.3.3.4, "Meteorological Instrumentation", with an alternate reference; (11) Allow the use of a containment atmosphere grab sampling capability as a backup to the hydrogen analyzers in TS 3.6.5.1, "Hydrogen Analyzers," and (12) incorporate additional reporting requirements in TS 6.9.2, "Special Reports".

The remaining issues associated with your applications dated February 22, 1985 and October 25, 1985 will be addressed in future correspondence.

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

Sincerely,

/s/

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

Enclosures:
1. Amendment No. 117 to DPR-53
2. Amendment No. 99 to DPR-69
3. Safety Evaluation

cc w/enclosure:
See next page

AT Subject to correction to references for T.S. 3.1.2.4 for Unit 1/2 in transmittal letter, SER & Notice. No issuance until expiration of notice period.

PBD#8	PBD#8	PBD#8	OELD
PMKreutzer	DJaffe	AThadani	
3/28/86	4/1/86	4/1/86	4/1/86

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

Calvert Cliffs Nuclear Power Plant

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 117
License No. DPR-53

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

   A. The applications for amendments by Baltimore Gas & Electric Company (the licensee) dated February 22, 1985 and October 25, 1985, comply 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 applications, 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. 117, 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: April 14, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 117

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
B 2-5	B 2-5
3/4 1-6	3/4 1-6
3/4 1-11	3/4 1-11
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
-	3/4 3-41a
-	3/4 3-42 (no change)
3/4 4-5	3/4 4-5
3/4 5-5	3/4 5-5
3/4 5-5a	3/4 5-5a
3/4 6-1	3/4 6-1
3/4 6-26	3/4 6-26
3/4 7-61a	3/4 7-61a
B 3/4 3-2	B 3/4 3-2
B 3/4 5-2	B 3/4 5-2
B 3/4 7-1	B 3/4 7-1
6-18a	6-18a

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.23 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.23 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to, or at least concurrently with, a safety injection.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 685 psia is sufficiently below the full-load operating point of 850 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of $\pm$ 85 psi which was based on the main steam line break event inside containment.

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the reactor coolant system will not exceed its Safety Limit. The specified setpoint in combination with the auxiliary feedwater actuation system ensures that sufficient water inventory exists in both steam generators to remove decay heat following a loss of main feedwater flow event.

Axial Flux Offset

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a DNBR of less than 1.23 nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore axial flux offset relationship.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.23.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, and the number of reactor coolant pumps operating. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.7×10^{-4} $\Delta k/k/^{o}F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER.

b. Less positive than 0.2×10^{-4} $\Delta k/k/^{o}F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.7×10^{-4} $\Delta k/k/^{o}F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

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

#See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER above 90% of RATED THERMAL POWER, within 7 EFPD after initially reaching an equilibrium condition at or above 90% of RATED THERMAL POWER.

c. At any THERMAL POWER, within 7 EFPD of reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 3% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE:

a. At least once per 18 months by verifying that each charging pump starts automatically upon receipt of a Safety Injection Activation Test Signal.

b. No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Above 80% RATED THERMAL POWER the two OPERABLE charging pumps shall have independent power supplies.

BORIC ACID PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1a above, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wide Range Neutron Flux	M	N.A.
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Coolant Cold Leg Temperature	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Level (Wide Range)	M	R
7. Steam Generator Pressure	M	R

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3.10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. As shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-10.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Deleted		
2. Containment Pressure	2	31
3. Wide Range Logarithmic Neutron Flux Monitor	2	31
4. Reactor Coolant Outlet Temperature	2	31
5. Deleted		
6. Pressurizer Pressure	2	31
7. Pressurizer Level	2	31
8. Steam Generator Pressure	2/steam generator	31
9. Steam Generator Level (Wide Range)	2/steam generator	31
10. Feedwater Flow	2	31
11. Auxiliary Feedwater Flow Rate	2/steam generator	31
12. RCS Subcooled Margin Monitor	1	31
13. PORV/Safety Valve Acoustic Flow Monitoring	1/valve	31
14. PORV Solenoid Power Indication	1/valve	31
15. Containment Water Level (Wide Range)	2	32, 33

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 31 - With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

ACTION 32 - With the number of OPERABLE post-accident monitoring channels one less than the minimum channel operable requirement in Table 3.3-10, operation may proceed provided the inoperable channel is restored to OPERABLE status at the next outage of sufficient duration.

ACTION 33 - With the number of OPERABLE post-accident monitoring channels two less than required by Table 3.3-10, either restore one inoperable channel to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4. The pressurizer shall be OPERABLE with a steam bubble and with at least 150 kw of pressurizer heater capacity capable of being supplied by emergency power. The pressurizer level shall be maintained within an operating band between 133 and 225 inches except when three charging pumps are operating and letdown flow is less than 25 GPM. If three charging pumps are operating and letdown flow is less than 25 GPM pressurizer level shall be limited to between 133 and 210 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water level shall be determined to be within the above band at least once per 12 hours.

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and

e. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psia.

   2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

   3. Verifying that a minimum total of 100 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

   4. Verifying that when a representative sample of 4.0 ± 0.1 grams of TSP from a TSP storage basket is submerged, without agitation, in 3.5 ± 0.1 liters of $77 \pm 10°F$ borated water from the RWT, the pH of the mixed solution is raised to ≥ 6 within 4 hours.

f. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

      a. High-Pressure Safety Injection pump.

      b. Low-Pressure Safety Injection pump.

SURVEILLANCE REQUIREMENTS (Continued)

g. By performing a flow balance test during shutdown following completion of HPSI system modifications that alter system flow characteristics and verifying the following flow rates for a single HPSI pump system*:

1. The sum of the three lowest flow legs shall be greater than 470** gpm.

h. By verifying that the HPSI pumps develop a total head of 2900 ft. on recirculation flow to the refueling water tank when tested pursuant to Specification 4.0.5.

* A HPSI pump system is a HPSI pump and one of two safety injection headers.
**These limits contain allowances for instrument error, drift or fluctuation.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.4.1.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. By verifying that the equipment hatch is closed and sealed, prior to entering Mode 4 following a shutdown where the equipment hatch was opened, by conducting a Type B test per Appendix J to 10 CFR Part 50.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$ (346,000 SCCM), 0.20 percent by weight of the containment air per 24 hours at P_a, 50 psig, or

2. $\leq L_t$ (61,600 SCCM), 0.058 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.

b. A combined leakage rate of $\leq 0.60\ L_a$ (207,600 SCCM), for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a (259,500 SCCM) or 0.75 L_t (46,200 SCCM), as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either P_a (50 psig) or at P_t (25 psig) during each 10-year service period.

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	$\leq$ 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 valves will be shut in MODES 1, 2, 3, and 4 per TS 3/4 6.1.7.

* 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 isolation valves is NA for MODES 1, 2, 3 and 4 per TS 3/4 6.1.7, during which time these valves must remain closed.

(4) Containment vent isolation valves shall be opened for containment pressure control, airborne radioactivity control, and surveillance testing purposes only.

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:

   1. Verify containment atmosphere grab sampling capability and prepare and submit a special report to the Commission pursuant to Specification 6.9.2 within the following 30 days, outlining the ACTION taken, the cause for the inoperability, and the plans and schedule for restoring the system to OPERABLE status, or

   2. 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 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
1-83-55	MAIN STEAM LINE ENCAPSULATION 27'	A	No	No
1-83-56	MAIN STEAM LINE ENCAPSULATION 27'	A	No	No
1-83-57	MAIN STEAM LINE ENCAPSULATION 27'	A	No	No
1-83-58	MAIN STEAM LINE ENCAPSULATION 27'	A	No	No
1-83-67	MAIN STEAM FROM S.G. #12 61'	I	Yes	No
1-83-69	MAIN STEAM FROM S.G. #12 61'	I	Yes	No
1-83-70	MAIN STEAM FROM S.G. #12 61'	I	Yes	No
1-83-71	MAIN STEAM FROM S.G. #12 61'	I	Yes	No
1-83-73	MSIV #11 HYDRAULIC SUPPLY 38'	A	No	No
1-83-74	MSIV #11 HYDRAULIC SUPPLY 38'	A	No	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
1-83-75	AUXILIARY STEAM ISOLATION VALVE BYPASS 32'	A	No	No
1-83-76	AUXILIARY FEED PUMP STEAM SUPPLY FROM S.G. #12 40'	A	No	No
1-83-76A	AUXILIARY FEED PUMP STEAM SUPPLY FROM S.G. #12 40'	A	No	No
1-83-77	AUXILIARY FEED PUMP STEAM SUPPLY FROM S.G. #12 40'	A	No	No

*Snubbers may be added to safey related systems without prior License Amendment to Table 3.7-4 provided that a revision to Table 3.7-4 is included with the next License Amendment request. Snubbers may be removed from safety related systems for the purpose of replacement by sway struts in accordance with the NRC's Safety Evaluation dated April 19, 1984, provided that a revision to Table 3.7-4 is included with the next License Amendment request.

**Modification to this table due to changes in high radiation areas shall be submitted to the NRC as part of the next License Amendment request.

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, as supplemented by Supplement 1 to NUREG-0737.

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.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The safety injection tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

Portions of the low pressure safety injection (LPSI) system flowpath are common to both subsystems. This includes the low pressure safety injection flow control valve, CV-306, the flow orifice downstream of CV-306, and the four low pressure safety injection loop isolation valves. Although the portions of the flowpath are common, the system design is adequate to ensure reliable ECCS operation due to the short period of LPSI system operation following a design basis Loss of Coolant Incident prior to recirculation. The LPSI system design is consistent with the assumptions in the safety analysis.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to $\geq$ 7.0. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The surveillance requirement for flow balance testing provides assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. Minimum HPSI flow requirements are based upon small break LOCA calculations which credit charging pump flow following an SIAS. Surveillance testing includes allowances for instrumentation and system leakage uncertainties. The 470 gpm requirement for minimum HPSI flow from the three lowest flow legs includes instrument uncertainties but not system check valve leakage. The OPERABILITY of the charging pumps and the associated flow paths is assured by the Boration System Specification 3/4.1.2. Specification of safety injection pump total developed head ensures pump performance is consistent with safety analysis assumptions.

3/4.5.4 REFUELING WATER TANK (RWT)

The OPERABILITY of the RWT as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

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

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

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

For two loop operation

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

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

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

where:

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

V = maximum number of inoperable safety valves per steam line

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

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

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

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

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

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. A capacity of 400 gpm is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 300°F when the shutdown cooling system may be placed into operation.

Flow control valves, installed in each leg supplying the steam generators, are set to maintain a nominal flow setpoint of 200 gpm plus or minus 10 gpm for operator setting band. The nominal flow setpoint of 200 gpm incorporates a total instrument loop error band of plus 25 gpm and minus 26 gpm for the motor-driven pump train. The corresponding values for the steam-driven pump train are plus 37 gpm and minus 40 gpm. The operator setting band, when combined with the instrument loop error, results in a total flow band of 164 gpm (minimum) and 235 gpm (maximum) for the motor-driven pump train. The corresponding values for the steam-driven pump train are 150 gpm (minimum) and 247 gpm (maximum). Safety analyses show that more flow during an overcooling transient and less flow during an undercooling transient could be tolerated; i.e., flow fluctuations outside this flow band but within the assumptions used in the analyses listed below, are allowable.

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

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Core Barrel Movement, Specification 3.4.11.

f. Fire Detection Instrumentation, Specification 3.3.3.7.

g. Fire Suppression Systems, Specifications 3.7.11.1, 3.7.11.2, 3.7.11.3, 3.7.11.4, and 3.7.11.5.

h. Penetration Fire Barriers, Specification 3.7.12.

i. Steam Generator Tube Inspection Results, Specification 4.4.5.5.a and c.

j. Specific Activity of Primary Coolant, Specification 3.4.8.

k. Containment Structural Integrity, Specification 4.6.1.6.

l. Radioactive Effluents - Calculated Dose and Total Dose, Specifications 3.11.1.2, 3.11.2.2, 3.11.2.3, and 3.11.4.

m. Radioactive Effluents - Liquid Radwaste, Gaseous Radwaste and Ventilation Exhaust Treatment Systems Discharges, Specifications 3.11.1.3 and 3.11.2.4.

n. Radiological Environmental Monitoring Program, Specification 3.12.1.

o. Radiation Monitoring Instrumentation, Specification 3.3.3.1 (Table 3.3-6).

p. Overpressure Protection Systems, Specification 3.4.9.3.

q. Hydrogen Analyzers, Specification 3.6.5.1.

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
License No. DPR-69

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

   A. The applications for amendments by Baltimore Gas & Electric Company (the licensee) dated February 22, 1985 and October 25, 1985 comply 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 applications, 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. 99 , 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: April 14, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 99

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
B 2-5	B 2-5
3/4 1-6	3/4 1-6
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
-	3/4 3-41a
-	3/4 3-42 (no change)
3/4 4-5	3/4 4-5
3/4 5-5	3/4 5-5
3/4 5-5a	3/4 5-5a
3/4 6-1	3/4 6-1
3/4 6-26	3/4 6-26
B 3/4 3-2	B 3/4 3-2
B 3/4 5-2	B 3/4 5-2
B 3/4 7-1	B 3/4 7-1
6-18a	6-18a

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.21 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.21 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to, or at least concurrently with, a safety injection.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 685 psia is sufficiently below the full-load operating point of 850 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of + 85 psi in the accident analyses which was based on the Main Steam Line Break event.

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the reactor coolant system will not exceed its Safety Limit. The specified setpoint in combination with the auxiliary feedwater actuation system ensures that sufficient water inventory exists in both steam generators to remove decay heat following a loss of main feedwater flow event.

Axial Flux Offset

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a DNBR of less than 1.21 nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore axial flux offset relationship.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.21.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, and the number of reactor coolant pumps operating. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.7×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than 0.2×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.7×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

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

#See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER above 90% of RATED THERMAL POWER, within 7 EFPD after initially reaching an equilibrium condition at or above 90% of RATED THERMAL POWER after each fuel loading.

c. At any THERMAL Power, within 7 EFPD of reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wide Range Neutron Flux	M	N.A.
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Coolant Cold Leg Temperature	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Level	M	R
7. Steam Generator Pressure	M	R

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. As shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-10.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		MINIMUM CHANNELS OPERABLE	ACTION
1.	Containment Pressure	2	31
2.	Wide Range Logarithmic Neutron Flux Monitor	2	31
3.	Reactor Coolant Outlet Temperature	2	31
4.	Pressurizer Pressure	2	31
5.	Pressurizer Level	2	31
6.	Steam Generator Pressure	2/steam generator	31
7.	Steam Genrator Level (Wide Range)	2/steam generator	31
8.	Auxiliary Feedwater Flow Rate	2/steam generator	31
9.	RCS Subcooled Margin Monitor	1	31
10.	PORV/Safety Valve Acoustic Flow Monitoring	1/valve	31
11.	PORV Solenoid Power Indication	1/valve	31
12.	Feedwater Flow	2	31
13.	Containment Water Level (Wide Range)	2	32, 33

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 31 - With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

ACTION 32 - With the number of OPERABLE post-accident monitoring channels one less than the minimum channel operable requirement in Table 3.3-10, operation may proceed provided the inoperable channel is restored to OPERABLE status at the next outage of sufficient duration.

ACTION 33 - With the number of OPERABLE post-accident monitoring channels two less than required by Table 3.3-10, either restore one inoperable channel to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

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

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4. The pressurizer shall be OPERABLE with a steam bubble and with at least 150 kw of pressurizer heater capacity capable of being supplied by emergency power. The pressurizer level shall be maintained within an operating band between 133 and 225 inches except when three charging pumps are operating and letdown flow is less than 25 GPM. If three charging pumps are operating and letdown flow is less than 25 GPM pressurizer level shall be limited to between 133 and 210 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water level shall be determined to be within the above band at least once per 12 hours.

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 100 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 4.0 $\pm$ 0.1 grams of TSP from a TSP storage basket is submerged, without agitation, in 3.5 $\pm$ 0.1 liters of 77 $\pm$ 10°F borated water from the RWT, the pH of the mixed solution is raised to $\geq$ 6 within 4 hours.

f. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

SURVEILLANCE REQUIREMENTS (Continued)

g. By performing a flow balance test during shutdown following completion of HPSI system modifications that alter system flow characteristics and verifying the following flow rates for a single HPSI pump system*:

   1. The sum of the three lowest flow legs shall be greater than 470** gpm.

h. By verifying that the HPSI pumps develop a total head of 2900 ft on recirculation flow to the refueling water tank when tested pursuant to Specification 4.0.5.

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* A HPSI pump system is a HPSI pump and one of two safety injection headers.
**These limits contain allowances for instrument error, drift or fluctuation.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.4.1.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. By verifying that the equipment hatch is closed and sealed, prior to entering Mode 4 following a shutdown where the equipment hatch was opened, by conducting a Type B test per Appendix J to 10 CFR Part 50.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$ (346,000 SCCM), 0.20 percent by weight of the containment air per 24 hours at P_a, 50 psig, or

2. $\leq L_t$ (44,600 SCCM), 0.042 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.

b. A combined leakage rate of $\leq 0.60\ L_a$ (207,600 SCCM) for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a (259,500 SCCM), or 0.75 L_t (33,400 SCCM), as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either P_a (50 psig) or at P_t (25 psig) during each 10-year service period.

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 valves will be shut in MODES 1, 2, 3 and 4 per TS 3/4 6.1.7.

* 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 isolation valves is NA for MODES 1, 2, 3 and 4 per TS 3/4 6.1.7, during which time these valves must remain closed.

(4) Containment vent isolation valves shall be opened for containment pressure control, airborne radioactivity control, and surveillance testing purposes only.

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:

   1. Verify containment atmosphere grab sampling capability and prepare and submit a special report to the Commission pursuant to Specification 6.9.2 within the following 30 days, outlining the ACTION taken, the cause for the inoperability, and the plans and schedule for restoring the system to OPERABLE status, or

   2. 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.

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, as supplemented by Supplement 1 to NUREG-0737.

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.

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The safety injection tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protection function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

Portions of the low pressure safety injection (LPSI) system flowpath are common to both subsystems. This includes the low pressure safety injection flow control valve, CV-306, the flow orifice downstream of CV-306, and the four low pressure safety injection loop isolation valves. Although the portions of the flowpath are common, the system design is adequate to ensure reliable ECCS operation due to the short period of LPSI system operation following a design basis Loss of Coolant Incident prior to recirculation. The LPSI system design is consistent with the assumptions in the safety analysis.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to $\geq$ 7.0. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The surveillance requirement for flow balance testing provides assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. Minimum HPSI flow requirements are based upon Small Break LOCA calculations which credit charging pump flow following a SIAS. Surveillance testing includes allowances for instrumentation and system leakage uncertainties. The 470 gpm requirement for minimum HPSI flow from the three lowest flow legs includes instrument uncertainties but not system check valve leakage. The OPERABILITY of the charging pumps and the associated flowpaths is assured by the Boration System Specifications 3/4.1.2. Specification of safety injection pump total developed head ensures pump performance consistent with safety analysis assumptions.

3/4.5.4 REFUELING WATER TANK (RWT)

The OPERABILITY of the RWT as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

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

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

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

For two loop operation

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

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

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

where:

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

V = maximum number of inoperable safety valves per steam line

U	=	maximum number of inoperable safety valves per operating steam line
106.5	=	Power Level-High Trip Setpoint for two loop operation
46.8	=	Power Level-High Trip Setpoint for single loop operation with two reactor coolant pumps operating in the same loop
X	=	Total relieving capacity of all safety valves per steam line in lbs/hour
Y	=	Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. A capacity of 400 gpm is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 300°F when the shutdown cooling system may be placed into operation.

Flow control valves, installed in each leg supplying the steam generators, are set to maintain a nominal flow setpoint of 200 gpm plus or minus 10 gpm for operator setting band. The nominal flow setpoint of 200 gpm incorporates a total instrument loop error band of plus 25 gpm and minus 26 gpm for the motor-driven pump train. The corresponding values for the steam-driven pump train are plus 37 gpm and minus 40 gpm.

The operator setting band, when combined with the instrument loop error, results in a total flow band of 164 gpm (minimum) and 235 gpm (maximum) for the motor-driven pump train. The corresponding values for the steam-driven pump train are 150 gpm (minimum) and 247 gpm (maximum). Safety analyses show that more flow during an overcooling transient and less flow during an undercooling transient could be tolerated; i.e., flow fluctuations outside this flow band but within the assumptions used in the analyses listed below, are allowable.

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

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

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Core Barrel Movement, Specification 3.4.11.

f. Fire Detection Instrumentation, Specification 3.3.3.7.

g. Fire Suppression Systems, Specifications 3.7.11.1, 3.7.11.2, 3.7.11.3, 3.7.11.4, and 3.7.11.5.

h. Penetration Fire Barriers, Specification 3.7.12.

i. Steam Generator Tube Inspection Results, Specification 4.4.5.5.a and c.

j. Specific Activity of Primary Coolant, Specification 3.4.8.

k. Containment Structural Integrity, Specification 4.6.1.6.

l. Radioactive Effluents - Calculated Dose and Total Dose, Specifications 3.11.1.2, 3.11.2.2, 3.11.2.3, and 3.11.4.

m. Radioactive Effluents -- Liquid Radwaste, Gaseous Radwaste and Ventilation Exhaust Treatment Systems Discharges, Specifications 3.11.1.3 and 3.11.2.4.

n. Radiological Environmental Monitoring Program, Specification 3.12.1.

o. Radiation Monitoring Instrumentation, Specification 3.3.3.1 (Table 3.3-6).

p. Overpressure Protection Systems, Specification 3.4.9.3.

q. Hydrogen Analyzers, Specification 3.6.5.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 117 AND 99

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 applications for license amendments dated February 22, 1985 and October 25, 1985, Baltimore Gas and Electric Company (BG&E) requested changes to the Technical Specifications for Calvert Cliffs Units 1 and 2. The proposed amendments would change the Unit 1 and Unit 2 TS to: (1) revise the Basis for the Containment Isolation Signal (CIS)/Safety Injection Actuation Signal (SIAS) setpoint for containment high pressure in TS Basis 2.2.1, "Reactor Trip Setpoints"; (2) change the allowable scheduling for moderator temperature coefficient (MTC) determination as required by TS 4.1.1.4.2c, "Moderator Temperature Coefficient"; (3) require that two charging pumps, required to be operable above 80% power, each be provided with an independent power supply per TS 3.1.2.4, "Charging Pumps - Operable" (Unit 1 only); (4) provide for additional channels associated with measurement of containment water level and change the statement regarding implementation of remedial actions in TS 3/4.3.3.6, "Post-Accident Instrumentation"; (5) correct a syntax error in TS 3.4.4, "Pressurizer" and a spelling error in TS 3/4.6.1.1., "Containment Integrity"; (6) update and clarify the reporting requirements of TS 6.9.2, "Special Reports"; (7) delete the Surveillance Requirements of TS 4.5.2g, "ECCS Subsystems T_{avg} [greater than or equal to] 300°F" - and redesignate the remaining Surveillance Requirements; (8) delete the reference to the 1971 Edition of the ASME Boiler and Pressure Vessel Code in TS Basis 3/4.7.1.1, "Safety Valves"; (9) delete a seismic sway arrester (snubber) from the operability and Surveillance Requirements of TS 3/4.7.8, "Snubbers" (Unit 1 only); (10) replace a reference in TS Basis 3/4.3.3.4, "Meteorological Instrumentation", with an alternate reference; (11) Allow the use of a containment atmosphere grab sampling capability as a backup to the hydrogen analyzers in TS 3.6.5.1, "Hydrogen Analyzers," and (12) incorporate additional reporting requirements in TS 6.9.2, "Special Reports."

This evaluation is partially responsive to the February 22, 1985 and October 25, 1985 applications. The remaining issues will be addressed in future correspondence.

DISCUSSION AND EVALUATION

BG&E has requested a change to TS Basis 2.2.1, "Reactor Trip Setpoints", with regard to the setpoint for reactor trip on containment pressure. The reactor protection system (RPS) for Calvert Cliffs Units 1 and 2 is an automatic safety system which trips the reactor when certain process variables exceed preset limits (setpoints). These setpoints are established to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. One such RPS setpoint would trip the reactor when containment pressure equals or exceeds 4 psig (per TS 2.2, "Limiting Safety System Settings",) and also initiate a containment isolation signal (CIS) to close key containment isolation valves. A separate containment pressure sensing system associated with the safety injection actuation signal (SIAS) initiates backup systems, including emergency core cooling systems, when containment pressure equals or exceeds 4.75 psig (per TS 3.3.2.1, "Engineered Safety Feature Actuation System Instrumentation").

Following the accident at TMI-2 the NRC established TMI Action Item II.E.4.2, "Containment Isolation Dependability", as set forth in NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. Action Item II.E.4.2 requires, in part, that licensees study their containment pressure history and establish a CIS setpoint that is "...reduced to a minimum compatible with normal operating conditions." The Basis for the containment pressure setpoint in TS Basis 2.2.1 states: "The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this is identical to the safety injection setpoint." A comparison of the RPS/CIS and SIAS setpoints in TS 2.2 and TS 3.3.2.1 indicates that there does not exist a requirement that these setpoints be "identical" or be initiated "concurrently." Accordingly, BG&E has requested that TS Basis 2.2.1 be reworded as follows: "The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to, or at least concurrently with, a safety injection." This wording is consistent with TS 2.2 and 3.3.2.1, meets the NRC staff objectives of TMI Action Item II.E.4.2, and is therefore acceptable.

BG&E has requested a change to TS 4.1.1.4.2c regarding the requirement to measure the end-of-cycle (EOC) moderator temperature coefficient (MTC). At the present time, the EOC MTC must be determined "...within 7 EFPD [effective full power days] after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm." BG&E has requested that the word "after" be changed to "of" in the above requirement to also allow measurement of the MTC up to 7 EFPD prior to reaching 300 ppm.

The Bases for TS 4.1.1.4.2 states the following, in part: "The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle." The need to measure the EOC MTC is associated with the need to confirm that this value will not be more negative than the most adverse value assumed in the safety analysis. Since the MTC becomes more negative near EOC due to decreasing soluble boron concentration, early measurement of the MTC is beneficial in predicting, and subsequently correcting, adverse trends. Since early determination of the MTC decreases the probability of accidents occurring with an excessively negative MTC, the proposed change to TS 4.1.1.4.2 is conservative and acceptable.

BG&E has proposed a change to TS 3.1.2.4, "Charging Pumps-Operation" which would require that: "Above 80% RATED THERMAL POWER the two OPERABLE charging pumps shall have independent power supplies." Recent loss-of-coolant accident (LOCA) analyses for Calvert Cliffs Units 1 and 2 have credited flow from the charging pumps in providing post-LOCA core cooling for LOCAs assumed to be initiated above 80% power. During the most recent Unit 2 LOCA review, it was noted by the NRC that the Calvert Cliffs charging pumps had not been required, in the TS, to have independent power supplies, as is implicit in the TS requirements for emergency core cooling system pumps, when these pumps are required to be operable. The NRC subsequently requested that BG&E propose such a TS requirement concerning charging pumps which was subsequently incorporated in Amendment 90 to Facility Operating License DPR-69, issued on November 21, 1985. BG&E subsequently proposed the same TS requirement for the Calvert Cliffs Unit 1 charging pumps.

The proposed change to TS 3.1.2.4 is consistent with existing safety analyses, provides additional protection with regard to diversity of power supply to the charging pumps, and is acceptable.

BG&E has proposed a change to TS 3.3.3.6, "Post-Accident Instrumentation", concerning the number of operable channels required for containment water level instrumentation. At the present time, TS 3.3.3.6 requires, in part, operability of a single channel of containment water level instrumentation. BG&E has proposed that a second channel of containment water level instrumentation be required to be operable. In addition, BG&E has proposed remedial action to be taken in the event that the additional channel of containment water level instrumentation becomes inoperable. While current remedial action requirements for a single inoperable containment water level monitor requires plant shutdown within 30 days should the instrumentation become inoperable, the remedial action requirement proposed for the additional containment water level instrumentation would only require its return to operation following an outage of sufficient duration to effect repairs. This proposed requirement notwithstanding, it is understood that other types of potential instrument failures might not require a power decrease or shutdown to effect repairs. These types of failures would be promptly corrected by the licensee. The combined effect on the TS of the additional required containment water level instrumentation, together with the proposed remedial action, would be to provide for operability of two instrument channels under most expected conditions since these channels have proven to be reliable to date. Should one instrument channel become inoperable, the proposed TS would revert to the existing TS, should the second instrument channel become inoperable, which would require plant shutdown within 30 days if the instrumentation cannot be returned to operable status. The first inoperable instrumentation would have to be restored to operable condition at the earliest possible time. We conclude that the proposed change to TS 3.3.3.6 would improve the required availability of containment water level instrumentation and is acceptable.

BG&E has proposed the correction of one syntax and one spelling error in the TS. The syntactic error appears in TS 3.4.4, "Pressurizer." The licensee has proposed deletion of the word "maximum" as it presently applies to both maximum and minimum values of allowable pressurizer level. The second proposed change to the TS would correct the spelling of the word "Operation" as it appears in TS 3/4.6.1, "Primary Containment." Correction of spelling and similar errors in the TS is administrative in nature and acceptable.

BG&E has requested a change to TS 6.9.2, "Special Reports" to add the following reporting requirements to this TS:

- O. Radiation Monitoring Instrumentation, Specification 3.3.3.1 (Table 3.3-6)

- P. Overpressure Protection Systems, Specification 3.4.9.3

While these reporting requirements are already incorporated in TS 3.3.3.1 and TS 3.4.9.3, they are proposed for incorporation in TS 6.9.2 to achieve consistency in the TS; therefore, the proposed change to TS 6.9.2 is acceptable.

BG&E has proposed deletion of TS 4.5.2g, "ECCS Subsystem-T_{avg} [greater than or equal to] 300°F." The purpose of this surveillance requirement is to verify proper setting of the emergency core cooling system (ECCS) throttle valves. The setting of the ECCS throttle valves had been a chronic problem at Calvert Cliffs. Due to a previously restrictive value of high pressure safety injection (HPSI) flow of 170 ± 5 gpm, the ECCS throttle valve settings had been required to be set at an intermediate opening position with a precision which exceeded the capability of the equipment. This situation resulted in a lack of repeatability for the settings. During the last reload for Calvert Cliffs Units 1 and 2, a combination of revised analysis and testing was presented in support of applications for license amendments which resulted in the elimination of the need for the restrictive HPSI flow requirement. The resulting TS changes, issued with License Amendments 104 and 90 for Calvert Cliffs Units 1 and 2, eliminated the need for ECCS throttle valve setting verification since these valves were now set to assume the full open position upon a safety injection actuation signal. The licensee had not requested deletion of TS 4.5.2g as part of their applications supporting License Amendments 104 and 90 due to an oversight.

The deletion of TS 4.5.2g provides consistency with the existing TS with regard to the crediting of flow from the charging pumps in the approved ECCS analysis, as reflected in the TS and the more liberal limits on HPSI flow verification per TS 4.5.2h. The proposed deletion of TS 4.5.2g is therefore acceptable.

BG&E has proposed deletion of the reference to the 1971 Edition of the ASME Code, Section XI, in TS Basis 3/4.7.1.1, "Safety Valves". This reference appears to be in error in that, by letter dated February 8, 1982, the NRC approved the use of the 1974 Edition of the ASME Code, Section XI, for inservice testing of pumps and valves at Calvert Cliffs Units 1 and 2. Moreover, TS 4.0.5 requires the licensee to use Section XI of the ASME Code for inservice inspection of pumps and valves in accordance with 10 CFR 50.55a(g) which specifies the correct edition of the code and addenda. Thus, deletion of the reference to the code edition provides consistency with existing TS 4.0.5 and is acceptable.

BG&E has proposed the deletion of a snubber from the operability and surveillance requirements of TS 3/4.7.8, "Snubbers." Table 3.7-4 of the TS contains a list of all snubbers which are the subject of operability and surveillance requirements. In addition, TS Table 3.7-4 contains the following provision for removal of snubbers from TS requirements: "Snubbers may be removed from safety related systems for the purpose of replacement by sway struts in accordance with the NRC's Safety Evaluation dated April 19, 1984 provided that a revision to Table 3.7-4 is included with the next License Amendment request." The licensee's October 25, 1985 request for license amendment informed the NRC that a snubber had been removed from Calvert Cliffs Unit 1 and replaced with a rigid sway strut. The deletion of the snubber from the TS is thus consistent with TS 3/4.7.8 and acceptable.

BG&E has proposed a change to TS Basis 3/4.3.3.4, "Meteorological Instrumentation." The proposed change would change the reference basis document as follows: "Regulatory Guide 1.23, Rev. 1 (Proposed), 'Meteorological Programs in Support of Nuclear Power Plants,' September 1980," would be replaced by "Regulatory Guide 1.23, 'Onsite Meteorological Programs,' February 1972, as supplemented by Supplement 1 to NUREG-0737." While existing and proposed references are equivalent, the latter consists of documents that have been approved by the NRC in final form and thus are more appropriate for reference in a TS Basis. Since both references are equivalent, the proposed change represents a change in nomenclature and thus is acceptable.

BG&E has proposed a change to the Calvert Cliffs TS to allow the use of containment atmosphere grab sampling as a back-up means of measuring containment hydrogen concentration. Current TS require plant shutdown within 30 days when one hydrogen analyzer becomes inoperable. The proposed TS change would allow for continued plant operation provided a grab sampling capability is demonstrated.

The prime function of the hydrogen analyzers is to monitor the containment atmosphere for hydrogen following a Loss-of-Coolant Accident (LOCA). The hydrogen analyzing system is common to both units. The system consists of two hydrogen analyzing subsystems, each consisting of a hydrogen analyzer cabinet, sample select cabinet, hydrogen sequencer panel, and remote hydrogen recorder. Each hydrogen analyzing subsystem can monitor the containment hydrogen concentration at six points, three in each containment structure. These locations have been selected to provide representative samples of the containment atmosphere. The sampling lines are run in groups of three through two separate containment penetrations.

During any period when one analyzer subsystem is out of service, the second subsystem is available to perform all necessary sampling evolutions. In addition, the availability of the hydrogen "grab sampling" capability provides added redundancy. A sample bomb equipped with a septum plug is located on the 45' level of the Auxiliary Building. The Laboratory Analyst can use a syringe to withdraw and subsequently analyze a containment atmosphere sample with the analyzer pump operating and the sequencer lined-up to the appropriate 135' containment elevation sample point. This capability has been demonstrated at Calvert Cliffs and represents a capability that provides a substantial backup for the installed analyzers.

The atmospheric grab sample capability incorporates design features, as enhanced by procedural controls, to prevent the uncontrolled release of containment atmosphere during sampling. The sample line from each containment contains two key (capture) lock actuated valves to prevent inadvertent operation. Down-stream of the key-lock valves and prior to the sample point, a manual isolation valve is provided which would allow the operator to quickly isolate the sample line in the event of anomalous conditions. Finally, the sample point is provided with a double septum to minimize leakage.

The proposed change to TS 3.6.5.1 would, upon inoperability of one hydrogen analyzer, allow continued plant operation provided that the grab sampling capability is verified within 30 days or the plant must be shut down. The licensee has also proposed that a special report be submitted within 30 days, "...outlining the ACTION taken, the cause for the inoperability, and the plans and schedule for restoring the system to OPERABLE status." This reporting requirement would also be incorporated in TS 6.9.2, "Special Reports," for consistency. The existing TS requirements specify that, in the event that one hydrogen analyzer becomes inoperable, the hydrogen analyzer must be made operable within 30 days or the plant must be shut down within 6 hours.

Since the atmospheric grab sample represents a hydrogen monitoring capability that is equivalent to an installed hydrogen analyzer, the operability of a single hydrogen analyzer with a grab sample backup capability would provide the same degree of redundancy as the operability of two hydrogen analyzers. Thus, the proposed changes to TS 3.6.5.1 and TS 6.9.2 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). These amendments also involve changes in recordkeeping, reporting or adminstrative 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.

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